NON-PUBLIC?: N
ACCESSION #: 8910310177
LICENSEE EVENT REPORT (LER)

FACILITY NAME: ST. LUCIE, UNIT 2 PAGE: 1 OF 5

DOCKET NUMBER: 05000389

TITLE: MANUAL REACTOR TRIP RESULTING FROM MULTIPLE DROPPED RODS DUE TO
UNRELATED EQUIPMENT FAILURES
EVENT DATE: 09/23/89 LER #: 89-007-00 REPORT DATE: 10/23/89

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 096

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: A. B. JOHNSON, SHIFT TECHNICAL ADVISOR

TELEPHONE: 407-465-3550

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AA COMPONENT: BKR MANUFACTURER: H141
X AA FU X999
X BA FCV L200
REPORTABLE NPRDS: Y
N
Y
SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 0558 on September 23, 1989, with St. Lucie Unit 2 in Mode 1 at 100% power, the unit experienced a dropped Control Element Assembly (CEA) in Regulating Group 1. The Reactor Control Operator (RCO) manually reduced main turbine load to match reactor power. At 0613, as the control rods were being inserted, four CEAs in Regulating Group 5, the lead bank, dropped in the core. The RCO immediately tripped the reactor from 96% power due to the multiple drop CEAs. The Standard Post Trip Actions were completed and the unit was quickly stabilized in Hot Standby, Mode 3.

The most probable cause of the event was a blown fuse causing the initial dropped CEA. The apparent cause for the four CEAs to drop was due to the tripping of the CEA Subgroup breaker. The root cause for the subgroup breaker trip has not been conclusively identified; however, testing the subgroup breaker revealed that the breaker tripped at a current less than designed.

The following corrective actions have been implemented: replaced the blown fuse and replaced the CEA Subgroup breaker.

END OF ABSTRACT

DESCRIPTION OF EVENT

At 0558 on September 23, 1989, with St. Lucie Unit 2 in Mode 1 at 100% power, the unit experienced a dropped Control Element Assembly (CEA)(EIIS:AA) in Regulating Group 1. CEA# 70 had dropped fully into the core. The Reactor Control Operator (RCO) manually reduced the main turbine load to match reactor power. The unit entered the Technical Specification ACTION statement 3.1.3.1.d for Movable Control Assemblies and had 63 minutes to realign CEA# 70 within 15 inches of other CEAs in its group or be in Hot Standby within 6 hours. The control room crew began to investigate the CEA# 70 drop and implemented the "CEA Off-Normal Operation And Realignment" procedure. The Assistant Nuclear Plant Supervisor (ANPS) went to the local Control Element Drive Mechanism Control System (CEDMCS)(EIIS:AA) panel and observed that the disconnect breaker for CEA# 70 had opened. The disconnect breaker for CEA# 70 was reclosed and the attempt to withdraw CEA# 70 was unsuccessful. The Nuclear Plant Supervisor (NPS) ordered a controlled power reduction due to an inoperable CEA. The RCO selected Manual Group 5 on CEDMCS and began inserting rods. At 0613, after inserting the control rods 2 to 3 inches, four CEAs in Regulating Group 5 dropped in the core. The ANPS ordered the RCO to manually trip the reactor and carry out the Standard Post Trip Actions. The reactor was immediately tripped from 96% power. The Steam Bypass Control System (SBCS)(EIIS:SB) operated to reduce primary average temperature (T-avg.) to the zero percent power setpoint of 532 degrees F. Auxiliary Feedwater (AFW)(EIIS:BA) actuated on low steam generator levels and functioned as designed with feedwater being supplied to both steam generators for Reactor Coolant System (RCS)(EIIS:AB) heat removal. The 2A AFW and 2B AFW pumps (electric motor driven) were supplying feedwater to the "A" and "B" steam generators, respectively. The 2C AFW pump (steam driven) also supplied feedwater to both the "A" and 'B" steam generators. The Standard Post Trip Actions were completed and the unit was quickly stabilized in Hot Standby, Mode

3.

The trip was an uncomplicated reactor trip with all safety functions verified as being met; however, some equipment had inadequate performance following the reactor trip. The RCO had to take manual control of PCV-8802, 10% steam bypass valve, because the valve did not automatically open. Subsequent to operator action, the SBCS functioned properly to maintain RCS heat removal. Approximately 45 minutes into the event, it was identified that the "A" AFW pump discharge valve, MV-09-9, did not respond to demand change from the control room nor could the valve be operated manually. MV-09-9 was mechanically bound such that AFW flow was approximately 200 gpm. The "B" AFW pump discharge valve, MV-09-10, limit switch position in the control room did not agree with feed flow response; i.e., MV-09-10 indicated full open but providing only 50% of maximum feed flow rate. The control room elected to locally operate MV-09-10. The 2C AFW pump, steam driven, was available to supply feedwater

to both steam generators via its own dedicated discharge valves, MV-09-11 & MV-09-12. The 2C AFW pump is designed with pump capacity equivalent to two electric motor driven AFW pumps. The SBCS and AFW problems did not impact the control room crew's ability to maintain the plant stable in Hot Standby, Mode 3. The cause of inadequate equipment performance will be discussed in the Analysis Of Event section.

CAUSE OF EVENT

The cause for the reactor trip was a manual reactor trip by the RCO following four dropped CEAs. The four CEAs that dropped were CEA# 69, CEA# 72, CEA# 75, and CEA# 78, which are powered from CEA Subgroup #18. The immediate cause of the four CEAs dropping was the tripping of CEA Subgroup #18 breaker. The root cause for the subgroup breaker trip has not been conclusively identified; however, testing of the subgroup breaker revealed that the breaker tripped at a current of approximately 30 amps, which is less than designed. The subgroup breaker is designed for a 40 amps continuous load. The CEA subgroup breaker was replaced.

The root cause for CEA# 70 dropping has not been conclusively identified. CEA# 70 is powered from CEA Subgroup #19. The most probable cause was a blown fuse in CEA Subgroup #19. This caused the CEA to lose one phase of the three phase power supply. The fuse was replaced for CEA Subgroup #19.

ANALYSIS OF EVENT

This event is reportable under 10 CFR 50.73 (a)(2)(iv), "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System."

Having one full length CEA drop followed by a full length CEA Subgroup drop is not an analyzed event in the St. Lucie Unit #2 Final Updated Safety Analysis Report. The Fuel Resources Group analyzed this event and determined that no Departure from Nucleate Boiling Ratio (DNBR) or Local Power Density (LPD) limits were violated at any time during this event. In addition, no Incore Neutron Detector alarms were received prior to the reactor trip.

The reactor trip was observed to be a routine manual reactor trip. The resulting transient was well enveloped by the St. Lucie Unit #2 Final Updated Safety Analysis Report.

PCV-8802 failed to automatically open on demand. PCV-8802 receives input from a T-avg. program to maintain RCS average temperature at 532 degrees F. The RCO transferred the individual controller of PCV-8802 to manual control for RCS heat removal. The failure

of the valve to auto open did not significantly affect the performance of the Steam Bypass Control System. In addition, the Atmospheric Dump Valves were available to provide an alternate means to accomplish the RCS heat removal safety function.

MV-09-9 lost the ability to be throttled due to mechanical binding in the valve's actuator after the RCO had throttled the discharge valve to a feed flow rate of 200 gpm. The cause for mechanical binding in the actuator was due to excessive wear of the upper stem threads leading up to galling and eventual binding between the upper stem and stem nut. MV-09-10 had erroneous limit switch position indication in the control room. The cause for the erroneous indication was the lack of engagement between the limit switch drive pinion and the drive sleeve bevel gear in the valve's actuator. The 2C AFW pump, steam driven, was available to supply feedwater to both steam generators via its own dedicated discharge valves, MV-09-11 & MV-09-12. The 2C AFW pump is designed with pump capacity equivalent to two electric motor driven AFW pumps.

From the analysis of the event, all Safety Functions were met and maintained; therefore, the health and safety of the public were not affected by this event.

CORRECTIVE ACTIONS

1. The fuse was replaced for CEA Subgroup #19.

2. CEA Subgroup #18 breaker was replaced. Future plans to test CEA Subgroup breakers during upcoming refueling outages.

3. Instrument & Control Department investigated the inadequate performance of the Steam Bypass Control System (SBCS) and did not find any problems with PCV-8802. The SBCS was tested and the system functioned properly. No cause could be found why PCV-8802 did not open automatically.

4. Mechanical Maintenance replaced the Limitorque operator upper stem and stem nut for MV-09-9. The actuator tested satisfactorily and was returned to service.

5.
Electrical Maintenance replaced the worn limit switch drive pinion gear for MV-09-10. The limit switch pinion gear and the drive sleeve bevel gear was verified for proper gear engagement. The actuator tested satisfactorily and was returned to service.

ADDITIONAL INFORMATION

FAILED COMPONENT:

Component: CEA Subgroup #18 Breaker, 250 VAC 50/60 Hz
Manufacturer: Heinemann Electric Company
Model #: AM3-A2A2A3-A

Component: Semiconductor Fuse
Manufacturer: International Rectifier
Model #: SF25X60

Component: MV-09-9 & MV-09-10 Actuators
Manufacturer: Limitorque
Model #: SMB-000

PREVIOUS SIMILAR EVENTS:

See LER #335-80-050 for previous manual reactor trip due to multiple dropped rods.

ATTACHMENT 1 TO 8910310177 PAGE 1 OF 1

P.O. Box 14000, Juno Beach, FL 33408-0420
FPL

OCTOBER 23 1989

L-89-381
10 CFR 50.73

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Re: St. Lucie Unit 2
Docket No. 50-389
Reportable Event: 89-07
Date of Event: September 23, 1989
Manual Reactor Trip Resulting From Multiple
Dropped Rods Due To Unrelated Equipment Failures

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR 50.73 to provide notification of the subject event.

Very truly yours,

D. A. Sager
Vice President
St. Lucie Plant

DAS/JRH/gmp

Attachment

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

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